Progress Energy

Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

Ref: 10 CFR 50.73

October 21, 2009
3F1009-04

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: LICENSEE EVENT REPORT 50-302/2009-003-00

Dear Sir:

Please find enclosed Licensee Event Report (LER) 50-302/2009-003-00. The LER discusses a manual reactor trip due to insertion of the Group 7 control rods caused by inadvertent contact of an inadequately protected (fused) test jumper to an unintended point. This report is being submitted pursuant to 10CFR50.73(a)(2)(iv)(A).

No new regulatory commitments are made in this letter.

If you have any questions regarding this submittal, please contact Mr. Dan Westcott, Superintendent, Licensing and Regulatory Programs at (352) 563-4796.

Sincerely,

J. W. Holt

James W. Holt
Plant General Manager
Crystal River Nuclear Plant

JWH/dwh

Enclosure

xc: Regional Administrator, Region II
Senior Resident Inspector
NRR Project Manager

IE22
NRR

NRC FORM 366 (9-2007) | U.S. NUCLEAR REGULATORY COMMISSION | APPROVED BY OMB: NO. 3150-0104 EXPIRES: 08/31/2010

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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4. TITLE
Manual Reactor Trip Due To Group 7 Control Rods Insertion Caused By Inadequately Protected Test Jumper

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
08	24	2009	2009	- 003 -	00	10	21	2009	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Dennis W. Herrin, Lead Engineer (Licensing and Regulatory Programs)	352-563-4633

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

At 11:00, on August 24, 2009, Progress Energy Florida, Inc. (PEF), Crystal River Unit 3 (CR-3) was operating in MODE 1 (POWER OPERATION) at 100 percent RATED THERMAL POWER when the Control Room staff received multiple alarms and observed the Group 7 control rods fully insert into the reactor core. The reactor was manually tripped prior to automatic actuation of the Reactor Protection System (RPS). Prior to this event, electricians were implementing Preventive Maintenance procedure PM-126, "Electrical Checks of CRD [Control Rod Drive] Power Train." When the Integrated Control System was placed in Automatic, the output driver within the Group 7 programmer caused an erroneous phase sequence to the control rod drive stators, culminating in inadequate magnetic force to restrain the rods from dropping during movement. The RPS responded as expected to the manual trip signal, control rods fully inserted and safety systems functioned as required. No reduction in the public health and safety was created. The programmer failure was caused by inadvertent test jumper contact while using an improperly fused test jumper. This caused an over-current failure of the output driver within the programmer. The programmer was replaced and PM-126 was placed on administrative hold. This report is submitted under 10CFR50.73(a)(2)(iv)(A). No previous similar occurrence has been reported to the NRC.

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EVENT DESCRIPTION

At 11:00, on August 24, 2009, Progress Energy Florida, Inc. (PEF), Crystal River Unit 3 (CR-3) was operating in MODE 1 (POWER OPERATION) at 100 percent RATED THERMAL POWER when the Control Room staff received multiple alarms and observed all eight of the Group 7 control rods [AA, ROD] fully insert into the reactor core [AC]. The reactor was manually tripped prior to automatic actuation of the Reactor Protection System (RPS) [JD]. Emergency Operating Procedure EOP-2, "Vital System Status Verification," was immediately entered before eventually transitioning to EOP-10, "Post-Trip Stabilization," at 11:28, on August 24, 2009.

The Group 7 control rods are the controlling group at CR-3. They are expected to insert and withdraw in small increments corresponding to Integrated Control System (ICS) [JA] commands when the ICS is in Automatic. Movement of the rods is supervised by a programmer/controller (programmer) [AA, PMC] dedicated to Group 7. It is a microcontroller based component which responds to commands from ICS (or alternately, manual control) by sequencing the firing of the six phases of the control rod drive stators, as needed, to provide motion. If no movement requests are initiated, the programmer will maintain two phases continuously energized to hold the rods in a fixed position. If the programmer does not provide any Silicon Controlled Rectifier (SCR) [AA, SCR] firing demand outputs, the rods will be released. The ICS signal is provided to the programmer via relays to move rods, depending on the command initiated.

Prior to this event, experienced electricians were in the process of implementing Preventive Maintenance procedure PM-126, "Electrical Checks of CRD [Control Rod Drive] Power Train." Shortly after completing Section 4.6, which performs electrical checks of the Group 7 regulating power supply, the Group 7 control rod drives were transferred back from the Auxiliary supply to their Normal supply and the ICS was placed back into Automatic. After being placed back in Automatic, the Neutron Error signal monitored by ICS approached the set-point, corresponding to an automatic rod movement demand. The expected response to a movement demand is that the ICS command would be received by the CRD command logic. It would then be passed on along with the speed ("run" speed when in Automatic) to the controlling group's (Group 7) programmer. The programmer should translate the input into the proper sequence before it is output to the Gate Drives which are used to fire the SCRs, resulting in repositioning the Group 7 rods to provide correction to the Neutron Error. When fired by the Gate Drives, the SCRs output 120VDC to their associated Control Rod Drive Stator coil. The programmer should fire SCRs in a 2-3-2-3 phase sequence for rod movement or simply maintain 2 phases steady for holding rods.

In this event, the first time rod movement was requested by ICS after transferring back to Automatic, the Group 7 control rods erroneously responded and dropped into the core. The deviation between the expected response and the actual response was that rather than moving incrementally in accordance with the received commands, the rods dropped completely into the core.

Upon initiation of the manual reactor trip, the main turbine [TA] automatically tripped and the 'A' and 'B' 4160V Unit Buses [EB, BUS] transferred from the Unit Auxiliary Transformer [EB, XMFR] to the Startup Transformer per design.

No structures, systems or components were inoperable at the start of the event that contributed to the event. No other pertinent maintenance or surveillance activities were in progress. Plant protection and non-protection systems operated normally during the manual reactor trip, with the exception of the following:

Four (4) Main Steam Safety Valves (MSSVs) [SB, RV] that opened required operator action to lower pressure in accordance with EOP-10 guidance to reseat post-trip.

When the programmer for Group 7 failed, Rod 7-5 indicated a slower drop response than the other rods in Group 7. (Nuclear Condition Report (NCR) 351744)

Pressure pulsations occurred in various Condensate System [SD] lines until Condensate Pump CDP-1B [SD, P] was re-coupled. (NCR 351741)

Manual actuation of the RPS is reportable to the NRC. At 12:48, on August 24, 2009, a non-emergency four-hour notification was made to the NRC Operations Center (Event Number 45286) in accordance with 10CFR50.72(b)(2)(iv)(B). This report is being submitted pursuant to 10CFR50.73(a)(2)(iv)(A).

SAFETY CONSEQUENCES

Manual actuation of the RPS occurred to shut down the reactor while the Main Feedwater System [SJ] maintained adequate Once-Through Steam Generator (OTSG) [SB, SG] levels. Upon initiation of the manual reactor trip, the RPS responded as expected, control rods fully inserted and safety systems functioned as required. No challenges to the RPS setpoints were identified. Both Main Feedwater Pumps [SJ, P] remained in operation throughout this event. No Emergency Feedwater Initiation and Control System [JB] actuation occurred or was required.

The event did not result in the release of radioactive material. No design safety limits were exceeded and no fission product barriers or components were damaged as a result. The manual reactor trip is bounded by the Final Safety Analysis Report accident analysis.

Based on the above discussion, PEF concludes that the RPS performed as designed and did not represent a reduction in the public health and safety. Since no loss of safety function occurred, this event does not meet the Nuclear Energy Institute (NEI) definition of a Safety System Functional Failure (reference NEI 99-02, Revision 6).

CAUSE

The unexpected drop of the Group 7 control rods was due to the failure of the programmer caused by inadvertent test jumper contact during PM-126, using an improperly fused test jumper. These two conditions caused an over-current failure of the output driver within the Group 7 CRD programmer, causing an erroneous phase sequence to the control rod drive stators, culminating in inadequate magnetic force to restrain the rods from dropping during movement. The improper placement of an improperly fused jumper is a combination of two

inappropriate acts. Since a proper fuse in the test jumper alone would have prevented the event, it is considered to be the root cause for this event.
PM-126 directs use of a fused jumper, with a current limit of 0.1 amp. The jumper fuse was checked and found to be a 1.0 amp fuse. This is not consistent with the procedure, and is not adequate to protect the associated equipment which has a maximum current rating of 0.5 amp. The jumper made inadvertent contact with a positive voltage/current source. It is feasible that the contact was momentary, and unknown to the worker, as the jumper may have simply brushed across an adjacent terminal on the way to the intended terminal.

CORRECTIVE ACTIONS

1. Surveillance Procedure SP-102, "Control Rod Drop Time Tests," was performed and demonstrated that control rod 7-5 will perform as required for plant shutdown conditions if the programmer releases the control rod normally. NCR 351744 was initiated.

2. A walkdown was performed on the Condensate System in accordance with Administrative Instruction AI-1701, "System Engineering Standards." No visible damage from the pressure pulsation was identified. NCR 351741 was initiated.

3. The programmer for the Group 7 control rods was replaced under Work Order 1609125-03.

4. An accountability session was conducted with personnel qualified to perform PM-126 and their Supervisors. Inadequate use of human performance tools regarding confirmation of jumper fuse rating and proper jumper placement for this event were discussed.

5. PM-126 has been placed on Administrative Hold pending human performance improvements.

6. Additional corrective actions are identified in NCR 351705.

PREVIOUS SIMILAR EVENTS

No previous similar events have been reported to the NRC.

ATTACHMENTS

Attachment 1 – Abbreviations, Definitions, and Acronyms
Attachment 2 – List of Commitments

Attachment 1

Abbreviations, Definitions, and Acronyms

AI	Administrative Instruction
CFR	Code of Federal Regulations
CR-3	Crystal River Unit 3
CRD	Control rod Drive
CDP	Condensate System Pump
EOP	Emergency Operating Procedure
ICS	Integrated Control System
MSSV	Main Steam Safety Valve
NCR	Nuclear Condition Report
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PEF	Progress Energy Florida, Inc.
PM	Preventive Maintenance
RPS	Reactor Protection System
SCR	Silicon Controlled Rectifier
SP	Surveillance Procedure
V	Volt
Vdc	Volts direct current

NOTES: Improved Technical Specification Defined terms appear capitalized in LER text {e.g., MODE 1}.

Defined terms/acronyms/abbreviations appear in parenthesis when first used {e.g., Reactor Building (RB)}.

EIIS codes appear in square brackets {e.g., reactor building penetration [NH, PEN]}

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Attachment 2

LIST OF COMMITMENTS

The following table identifies those actions committed by PEF in this document. Any other actions discussed in the submittal represent intended or planned actions by PEF. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Superintendent, Licensing and Regulatory Programs of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	DUE DATE
No new regulatory commitments are contained in this submittal.	N/A